Dated at Rockville, Maryland, this 22nd day of February 1995.

For the Nuclear Regulatory Commission.

Ronald W. Hernan,

Acting Director, Project Directorate I–4, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–4975 Filed 2–28–95; 8:45 am]

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[Docket No. 50–413]

Duke Power Company, et al., Catawba Nuclear Station, Unit No. 1; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from certain requirements of 10 CFR Part 50, Appendix J, Paragraph III.D.1.(a), Type A Tests, to the Duke Power Company, et al. (the licensee), for operation of the Catawba Nuclear Station, Unit No. 1, located in York County, South Carolina, in accordance with Facility Operating License No. NFP–35.

Environmental Assessment

Identification of the Proposed Action

This Environmental Assessment has been prepared to address potential environmental issues related to the licensee's application of October 18, 1994, as supplemented on February 7, 1995. The proposed action would exempt the licensee from the requirements of 10 CFR Part 50, Appendix J, Paragraph III.D.1.(a), to the extent that a one-time schedular extension would permit rescheduling the third containment integrated leak rate test (ILRT) in the first 10-year service period from the end-of-Cycle 8 outage until the end-of-Cycle 9 outage. The requested exemption would also allow the decoupling of this third test from the endpoint of the first 10-year inservice inspection.

The Need for the Proposed Action

The current containment integrated leakage rate (ILRT) requirements for Catawba Units 1 and 2, pursuant to Appendix J, are that, after the preoperational leak rate test, a set of three Type A tests must be performed at approximately equal intervals during each 10-year period. Also, the third test of each set must be conducted when the plant is shut down for the 10-year plant inservice inspection. This is reflected in the Catawba Technical Specifications (TS) as a testing interval of once each 40 months plus or minus 10 months, for a frequency of three times in a 120-month period. To date, for Catawba Unit 1, the preoperational and the first two periodic ILRTs have been conducted. The most recent ILRT was conducted in March 1991, approximately 47 months ago. Thus, in accordance with Appendix J and the current TS, and ILRT would have to be conducted during the refueling outage beginning in February 1995 (the end-of-cycle (EOC) 8 outage).

The licensee has requested an exemption from Appendix J and a corresponding change to the TS that would allow a one-time change to the interval for the Unit 1 ILRT from 40 plus or minus 10 months to 60 plus or minus 10 months (once each 5 years). This would allow the EOC–8 ILRT to be rescheduled for EOC–9. Therefore, the need for the licensee's proposed action is to allow a longer interval between the Catawba Unit 1 second and third periodic Type A ILRTs which will result in a cost savings to the licensee.

Environmental Impacts of the Proposed Action

The proposed one-time exemption would not increase the probability or consequences of accidents previously analyzed and the proposed one-time exemption would not affect facility radiation levels or facility radiological effluents. The licensee has analyzed the results of previous Type A tests performed at the Catawba Nuclear Station, Unit No. 1. The licensee has provided an acceptable basis for concluding that the proposed one-time extension of the Type A test interval would maintain the containment leakage rates within acceptable limits. Accordingly, the Commission has concluded that the one-time extension does not result in a significant increase in the amounts of any effluents that may be released nor does it result in a significant increase in individual or cumulative occupational radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed exemption.

With regard to potential nonradiological impacts, the proposed exemption only involves Type A testing on the containment. It does not affect nonradiological plant effluents and has no other environmental impact. Accordingly, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed exemption.

Alternatives to the Proposed Action

Since the Commission has concluded there is no measurable environmental impact associated with the proposed exemption, any alternatives with equal or greater environmental impact need not be evaluated. The principal alternative to this action would be to deny the request for exemption. Such action would not reduce the environmental impacts of plant operations.

Alternative Use of Resources

This action does not involve the use of resources not previously considered in the ''Final Environmental Statement Related to the Operation of Catawba Nuclear Station Unit No. 1,'' dated January 1983.

Agencies and Persons Consulted

In accordance with its stated policy, the NRC staff consulted with the South Carolina State official regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed exemption.

For further details with respect to this action, see the licensee's letter dated October 18, 1994, as supplemented February 7, 1995, which are available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the York County Library, 138 East Black Street, Rock Hill, South Carolina.

Dated at Rockville, Maryland, this 23rd day of February 1995.

For the Nuclear Regulatory Commission.

Herbert N. Berkow,

Director, Project Directorate II–3, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–4976 Filed 2–28–95; 8:45 am]

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Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Pubic Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission on NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the

Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 3, 1995, through February 16, 1995. The last biweekly notice was published on February 15, 1995 (60 FR 8739).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By March 31, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shale be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to *(Project Director)*: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)(v) and 2.714(d).

For further details with respect to this section, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, Maricopa County, Arizona

Date of amendment requests: December 7, 1994.

Description of amendment requests: The proposed amendment would revise the capacity of the ultimate heat sink (UHS) as described in the bases of Technical Specification 3/4.7.5, ''Ultimate Heat Sink,'' from providing a 27-day cooling water supply to providing a 26-day cooling water supply. In addition, the reference to Regulatory Guide 1.27 in the bases of this TS would also be revised to reference the January 1976 revision rather than the March 1974 revision.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensees have provided their analysis about the issue of no significant hazards consideration, which is presented below:

Standard 1—Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The Essential spray pond system and the UHS do not initiate any accidents in Chapters 6 or 15 of the UFSAR [Updated Final Safety Analysis Report]. The justification and basis for the time that the UHS is available is not changed and continues to be consistent with the guidance in Regulatory Guide 1.27. The existing Technical Specification requirements and those components to which they apply are not altered by this Technical Specification amendment. Therefore, the change to the bases for Technical Specification 3/4.7.5 does not increase the probability of occurrence or the consequences of any previously evaluated accident.

Standard 2—Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The requirements for Technical Specification 3/4.7.5 are not changed. This amendment has no impact on plant maintenance, testing, shutdown equipment, or component qualification. Therefore, the possibility of a new or different kind of accident is not created by this amendment.

Standard 3—Does the proposed change involve a significant reduction in a margin of safety?

The change to the bases for Technical Specification 3/4.7.5 does not significantly alter existing Technical Specification requirements or those coponments to which they apply. The justification and basis for the time that the UHS is available without makeup is not changed and continues to be consistent with the guidance in Regulatory Guide 1.27. Regulatory Guide 1.27 states that ''A capacity less than 30 days may be acceptable if it can be demonstrated that replenishment can be effected to ensure that continuous capability of the sink to perform its safety functions, taking into account the availability of replenishment equipment and limitations that may be imposed on ''freedom of movement'' following an accident.'' This change does not effect the continuous capability of the UHS to perform its safety function of providing decay heat removal capability following an accident. The change updates the design basis of the UHS using more realistic conditions based on plant experience. Therefore, the change in the capacity of the UHS without makeup from 27 days to 26 days will not involve a significant reduction in margin of safety for the ultimate heat sink.

The NRC staff has reviewed the licensees' analysis and, based on that review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Attorney for licensees: Nancy C. Loftin, Esq., Corporation Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Project Director: Theodore R. Quay.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: June 18, 1992, as supplemented December 8, 1992, and revised February 3, 1995.

Description of amendment request: The proposed Technical Specification (TS) amendment adds limiting conditions of operation and surveillance requirements for the pressurizer power-operated relief valves (PORVs) and their associated block valves whenever average temperature (Tavg) is above 350 degrees F or the reactor is critical. Specifications have also been added for low-temperature overpressure protection whenever Tavg is less than 350 degrees F and the reactor coolant system is not vented to the containment. The February 3, 1995, revision made editorial changes to previous TS pages and made changes to conform with an additional provision of the guidance for surveillance testing of the block valves associated with the pressurizer PORVs. In addition, the licensee has requested an editorial change to TS page 3.1.–11 to revise the references to two figures that have been superseded.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The requested revision does not involve a significant increase in the probability or consequences of an accident previously

evaluated. The proposed revision to our previous Technical Specification (TS) change request dated June 18, 1992, would help assure the availability of the block valves for accident mitigation. The availability of the block valves for accident mitigation has been found to outweigh any negative safety consequences associated with full cycle testing of a block valve isolating a pressurizer power-operated relief valves (PORV) with ''excessive'' seat leakage. There would be no significant increase in the probability or consequences of an accident previously evaluated since this event is fully bounded by the failing open of a single pressurizer code safety relief valve event which is analyzed in Chapter 15 of the Updated Final Safety Analysis Report. Accordingly, the requested revision will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The requested revision to our previous TS change request does not create the possibility of a new or different kind of accident from any accident previously evaluated. Periodic testing of the block valves in accordance with the requested revision is only intended to assure the functioning and capability of the block valves. The requested revision will only clarify the conditions when block valve surveillance testing is required. The performance of this testing is intended to improve block valve availability and thereby assure the capability of certain accident mitigation strategies identified within Abnormal and Emergency Operating Procedures. Therefore, the requested revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The requested revision to our previous TS change request does not involve a significant reduction in the margin of safety. The requested revision is intended to help assure block valve availability to support certain accident mitigation strategies. This additional assurance of block valve availability and functioning increases the margin of safety. Accordingly, the requested revision will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: William H. Bateman.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: December 14, 1994.

Description of amendment request: The proposed amendments would revise technical specifications related to allowed outage times (AOT) and surveillance test intervals (STI) for certain actuation instrumentation in the reactor protection system (RPS), primary containment isolation system (PCIS), emergency core cooling system (ECCS), recirculation pump trip, reactor core isolation cooling (RCIC), control rod withdrawal block, monitoring, and feedwater/main turbine trip systems. These changes are generally consistent with General Electric topical reports which have been reviewed and approved by the NRC. The changes also include revising the Feedwater/Main Turbine Trip LCO 3.3.8 action statement to achieve consistency with existing instrumentation LCOs; deleting the surveillance of the APRM Neutron Flux—High, Setdown functional unit in Operational Condition 1; revising the applicability of the provisions of Specification 4.0.4 to several Reactor Protection System and Control Rod Withdrawal Block Instrumentation surveillance requirements; adding the requirement to perform shiftly channel checks for applicable RPS, PCIS, ECCS, and RCIC instrumentation channels equipped with master trip units; and other changes to correct typographical errors and to delete cycle specific footnotes which are no longer applicable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

It has been determined that the changes do not constitute a Significant Hazards Consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

a. The proposed changes increase the STI and AOT for actuation instrumentation supporting RPS, ECCS, Isolation, CRBF, RCIC, ATWS-RPT, EOC-RPT, Monitoring, and Feedwater/Main Turbine Trip System Actuation functions. There are no changes in instrumentation configuration and function, and no instrumentation setpoints are changed. Because of this there is no change in the probability of occurrence of an accident or the consequences of an accident or the consequences of malfunction of equipment. With respect to the probability of equipment malfunction, topical reports prepared by GE demonstrate that there is a reduction in scram frequency for the RPS, but in the case of the ECCS there is a small increase in the unavailability of the water injection function. This increase in unavailability was judged acceptable by GE. The NRC concurred with this conclusion in its review and approval of the topical reports. The proposed changes are consistent with the Safety Evaluation Reports issued for the topical reports.

b. The changes proposed for the Feedwater/Main Turbine Trip LCO action statements provide actions which are consistent with presently existing instrumentation LCOs. The design and function of the feedwater/main turbine trip instrumentation to trip the feedwater pumps and the main turbine upon detection of a Level 8 event is not altered. The probability and/or consequences of this moderate frequency transient are not increased.

c. The APRM Neutron Flux—High, Setdown scram setting provides adequate thermal margin between the setpoint and the safety limits for operation at low pressure and low flow during a plant startup. This function remains in effect until the mode switch is placed in the Run (Operational Condition 1) position, at which time it is bypassed. Deleting the requirement for the surveillance of the APRM Neutron Flux—High, Setdown functional unit in Operational Condition 1 is appropriate since its function is not applicable in this mode. This deletion serves to achieve consistency between Technical Specification Tables and the Bases section.

d. The changes associated with Specification 4.0.4 are administrative in nature and are intended to provide the plant operators with better guidance for its application. In cases where complete surveillances cannot be achieved, such as during a plant shutdown, then the required surveillances will be performed within 24 hours of entering the Mode or condition in which the surveillance is required. The stabilization of the plant will be of primary consideration. This change does not affect the evaluation for any accident presented in Chapter 15 of the UFSAR. The APRM Fixed Neutron Flux—High quarterly functional tests most of the APRM channel equipment associated with the APRM Neutron Flux—High, Setdown scram.

Additionally, the expected result of the functional tests associated with the SRMs, IRMs, and APRMs is to demonstrate the operability of the instrumentation. Therefore, 24 hours is a reasonable time to permit the surveillances to be performed upon entering the mode or condition in which the surveillance is required.

e. The proposal to include the performance of channel checks as requirements of technical specifications is administrative in nature. Presently, channel checks performed for the applicable analog instrumentation in reactor vessel water level applications is controlled solely by procedure. Adding this requirement to the technical specifications provides for the appropriate controls of the surveillances, above and beyond that presently controlled by procedure.

f. The proposed administrative changes are offered to correct typographical errors and delete cycle specific footnotes which are no longer applicable. The nature of the changes precludes them from impacting previously analyzed accidents.

The proposed changes therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

a. The proposed changes increase the STI and AOT for certain actuation instrumentation in the RPS, ECCS, Isolation, CRBF, RCIC, ATWS-RPT, EOC-RPT, Monitoring, and Feedwater/Main Turbine Trip systems. There are no changes in instrumentation configuration and function, and no instrumentation setpoints are changed.

b. The changes to the Feedwater/Main Turbine Trip LCO action statements allow the plant operators a maximum degree of operational flexibility, while maintaining the instrumentation and protection needed for terminating the feedwater controller failure transient. The single failure proof criterion of the level sensors is maintained, and the logic of the protective instrumentation is not compromised. The changes to the LCO action statements do not constitute a change to the facility or its operation as described in the Safety Analysis Report.

c. Deleting the requirement for surveilling the APRM Neutron Flux—High, Setdown functional unit in Operating Condition 1 does not degrade thermal margins. The margin accommodates the anticipated maneuvers associated with plant power ascension. During a plant shutdown, rod insertion maneuvers, recirculation flow reduction, and xenon build-in all contribute to negative reactivity insertion which precludes the degradation and violation of thermal margins. The functions of the APRMs required to be OPERABLE in Operational Condition 1 which are in effect remain to ensure that reactor core thermal margins are not compromised.

d. The conduct of neutron instrument functional tests in the plant mode or condition in which the trips are applicable eliminates unnecessary testing during normal plant operations. The expected result of the functional testing is to demonstrate the operability of the instruments. The failure of any single instrument channel will neither cause nor prevent either a reactor scram or a control rod block.

e. Including the performance of channel checks for the applicable analog instrumentation as part of the technical specifications transfers control of the required surveillances from procedure to the technical specifications, as appropriate. The administrative nature of this change does not alter the functions, setpoints, or configuration of the associated instrumentation.

f. The administrative nature of the changes prevents them from affecting the functions, setpoints, or configuration of the associated instrumentation from being affected by the changes.

The proposed changes do not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

(3) Involve a significant reduction in the margin of safety because:

a. Setpoints are based upon the drift occurring during an 18 month calibration interval. The bases in the Technical Specifications either do not discuss STI, or state ''* * * one channel may be inoperable for brief intervals to conduct required surveillance.'' The proposed changes are bounded by the analyses of the topical reports. These analyses, which were prepared by GE and approved by the NRC, examined the effects of extending STI and AOT and found that the proposed changes would not involve a significant reduction in the margin of safety.

b. The proposed changes to the turbine trip LCO action statements do not change any of the settings of the Level 8 setpoints. The single failure criteria of the multiple level sensors which sense and detect the Level 8 setpoint remains intact. The LCO maintains the requirement that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Level 8 signal. Scram trip signals from the turbine retain the design feature that a single failure will neither initiate nor impede the initiation of a reactor scram (trip).

c. The setting, function, and conditional requirements of the APRM Neutron Flux—High, Setdown function are not altered. This change serves to achieve consistency between two Technical Specifications Tables. This eliminates the need for surveilling a function in a mode which is not applicable. The functions of the APRMs required to be OPERABLE in Operational Condition 1 remain to ensure that reactor core thermal margins are not compromised.

d. The reference to 4.0.4 applicability will assist to ensure consistent interpretation of the technical specifications by the plant operators. This assists in ensuring that the plant is operated within technical specification limitations. This change does not affect trip instrumentation setpoints, and the scram function of the RPS is assured by the weekly functional testing of the Manual Scram.

e. Including the instrumentation channel checks as part of technical specification requirements provides an appropriately regimented method of controlling the conduct of the surveillances. None of the functions, setpoints, or configuration of the associated analog instrumentation is affected by this administrative change.

f. The administrative nature of the changes serves to provide more concise guidance to the plant operating staff, and as such do not impact the safety margin.

The proposed changes do not significantly reduce the margin of safety as defined in the basis for any Technical Specification.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

NRC Project Director: Robert A. Capra.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: January 13, 1995.

Description of amendment request: The proposed amendments would revise the pressure alarm setpoint allowable values for the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system ''keep filled'' pressure instrumentation channels. The purpose of the proposed change is to lower the setpoint allowable values for these parameters to more realistic values based upon calculations performed by the licensee reflecting design changes and system performance. Also, the term ''setpoint'' is being changed to ''setpoint allowable value'' to clarify the use of the values. Additionally, two administrative/editorial changes are included to delete technical specification footnotes which are no longer applicable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

a. The proposed change in the technical specification allowable values for the ECCS and RCIC discharge line ''keep filled'' alarm instrument channels does not change the design bases or function of these systems as described in the technical specifications and UFSAR. An analysis performed by engineering demonstrates that the proposed allowable values are sufficient for verifying that the ECCS and RCIC pump discharge lines are full of water. In addition, setpoint calculations have been performed to verify that sufficient margin exists between the recommended calibration setpoints and the analytical limits for these instrument channels to account for all applicable instrument errors. This provides high assurance that the trip setpoints of these instrument channels will not drop below the minimum required value. The ''keep filled'' instrumentation is not a factor in the assumptions of any accidents, thus, the probability of analyzed accidents is not increased.

b. The proposed technical specification amendment does not revise the configuration of the ECCS and RCIC discharge line ''keep filled'' instrument channels or sensing lines. The proposed setpoint allowable values and associated calibration setpoints are within the calibration ranges of the existing pressure switches. Thus, implementation of the proposed amendment does not involve any physical alterations to the plant except for the recalibration of the pressure switches to the new calibration setpoints.

c. The ECCS and RCIC discharge line ''keep filled'' instrument channels only perform a monitoring function. Other than ensuring system readiness they do not perform a function important to safety. Thus, the probability of a ECCS or RCIC failure is not increased since the operation and function of the ECCS and RCIC discharge line fill systems is not affected by this change.

d. The failure of a ECCS or RCIC discharge line fill system will not go undetected by the proposed change, since water leg pump trips are annunciated in the control room. In addition, quarterly surveillances are performed on these pumps to check for degradation.

e. The ECCS and RCIC discharge line fill systems are not used to mitigate the consequences of an accident or transient. These systems are not required after the ECCS and RCIC pumps are activated.

Therefore, the proposed change does not cause an increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because: This technical specification amendment only lowers the trip setpoint allowable values for the ECCS and RCIC discharge line ''keep filled'' alarm instrumentation channels. As described above, the proposed setpoint allowable values are sufficient for verifying that the ECCS and RCIC discharge lines are full of water. Thus, the probability of a water hammer occurring during system activation for a surveillance test is not increased. In addition, each instrument channel is independent from the other channels so that a failure in one channel will not propagate to another channel. Therefore, the operation of the facility in accordance with the proposed amendment does not create the possibility of a new or different kind of accident.

(3) Involve a significant reduction in the margin of safety because: The margin of safety is not affected by this amendment, because this change involves monitoring instrumentation only. The purpose of the ECCS and RCIC discharge line ''keep filled'' alarms is to alert the operators when a ECCS or RCIC system may not be operable due to empty or partially empty discharge lines. The proposed amendment does not alter or degrade this function, since the new setpoint allowable values are adequate for verifying that the discharge lines are full of water. Therefore the operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

NRC Project Director: Robert A. Capra.

Consumers Power Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: January 13, 1995

Description of amendment request: The proposed amendment would modify the required settings, and allowable ''as found'' and ''as left'' tolerances for the primary and secondary safety valves. The proposed limits would allow installed primary and secondary valve settings to be within a 3% tolerance of their nominal settings, but would require returning the valve settings to within 1% of the nominal settings if the valves are removed from the piping for maintenance or testing.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The following evaluation supports the finding that operation of the facility in accordance with the proposed technical specification change would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed change to the Technical Specifications increases the acceptable as found tolerance for the pressurizer safety valves. The most limiting overpressure event, loss of external load, has been analyzed to account for this change. The loss of external load analysis was performed using a conservative 25% steam generator tube plugging and an initial pressurizer level of 67.8% (providing an approximate 10% conservative margin above programmed pressurizer level for full power). Primary and secondary safety valve accumulation was conservatively accounted for and the setpoint tolerance of +3% was assumed. Reactor trip on turbine trip was assumed to be disabled and the atmospheric dump valves were assumed unavailable. The results of the analysis demonstrated primary and secondary system pressures within 110% of design pressures. Therefore, the consequences of overpressurization events will not be significantly increased with a +3% tolerance on the primary safety valve setpoints. The proposed Technical Specifications change will not affect normal plant operation and will not increase the probability of an accident.

A review of all DNB [departure from nucleate boiling] analyses was performed to ensure that predicted pressurizer pressures for those analyses would not be affected by a -3% tolerance on the lowest setpoint valve. The DNB analyses for which significant primary system pressure increases were predicted do not result in pressures high enough to lift the pressurizer safety valves with the proposed tolerance. A conservative DNB analysis that bounds the consequences of inadvertent opening of a pressurizer safety valve has also been previously performed with predicted acceptable results. If a pressurizer safety valve were to stick open, the consequences would be bounded by the small break LOCA [loss-of-coolant accident] analysis. Therefore, the consequences due to a -3% tolerance on the primary safety valve setpoints will not increase the consequences or probability of an accident.

The proposed revision removes the requirement for one operable pressurizer safety valve to be installed whenever the reactor head is on the vessel. Instead, proposed Specification 3.1.7.1 requires all pressurizer safety valves to be operable above cold shutdown, and overpressure protection during cold shutdown is provided by existing Specification 3.1.8.2, Power Operated Relief Valves.

The proposed Technical Specifications change also lists the lift settings for each of the primary and secondary system safety valves. This change will not affect the operation or function of the valves. Therefore, the probability and consequences of previously evaluated accidents will not be increased.

2. *Create the possibility of a new or different kind of accident from any previously evaluated.*

The proposed changes to Technical Specifications will not affect the manner in which the plant operates. The proposed increase in pressurizer safety valve lift setting tolerance could change the pressure at which the valves open in an overpressurization event, but would not create the possibility of a new or different kind of accident. Since Technical Specification 3.1.8 addresses primary system overpressurization during cold shutdown, the proposed removal of the requirement for an operable pressurizer safety valve to be installed whenever the reactor head is on the vessel will not create the possibility of a new overpressurization event during cold shutdown. The proposed change to list the lift settings for the individual primary and secondary safety valves will have no effect on the safety function of the valves. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Involve a significant reduction in a margin of safety.*

The proposed changes to Technical Specifications do not affect the DNB analyses that have been previously performed. The most limiting overpressurization event, loss of external load, has been conservatively analyzed accounting for the proposed changes and demonstrated that the primary and secondary system pressures remain within 110% of the design pressures. Overpressurization during cold shutdown is addressed by Technical Specification 3.1.8. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: John N. Hannon.

Consumers Power Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan.

Date of amendment request: February 10, 1995.

Description of amendment request: The proposed amendment would modify the Technical Specifications to allow a one time deferral of several 18-month interval surveillance tests until the upcoming scheduled refueling outage to avoid the necessity of imposing a plant shutdown solely for the sake of their performance.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The following evaluation supports the finding that operation of the facility in accordance with the proposed Technical Specifications (TS) would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated.*

Deferring surveillance testing will introduce no new operating conditions, change no equipment operating procedures, and change no plant systems or equipment. Therefore, operation of the facility in accordance with the proposed TS would not result in a significant increase in the probability of an accident previously evaluated.

Deferring surveillance testing of snubbers and instrument channels could allow minor degradations of snubber condition or small changes in instrument setpoints or calibration to progress some amount beyond that point which would occur with a shorter surveillance interval. A review of the recent test history for the subject surveillance indicates that no significant snubber degradation or instrument drift was found. It is not expected that, even with the proposed surveillance deferral, snubber conditions or instrument settings will be found to exceed conditions allowable by the Technical Specifications. Therefore, operation of the facility in accordance with the proposed TS would not result in a significant increase in the consequences of an accident previously evaluated.

2. *Create the possibility of a new or different kind of accident from any previously evaluated.*

Deferring surveillance testing will introduce no new operating conditions, change no equipment operating procedures, and change no plant systems or equipment. Therefore, operation of the facility in accordance with the proposed TS would not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Involve a significant reduction in a margin of safety.*

A review of past performance of the subject surveillance tests indicate that the requested deferral of testing would not have a significant effect on the results of the tests when they are performed prior to the startup for cycle 12. Most of the affected instrumentation is monitored each shift by channel checks, which would disclose major failures or significant drift. Therefore, operation of the facility in accordance with the proposed TS would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: John N. Hannon.

Duke Power Company, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: November 2, 1994.

Description of amendment request: The proposed amendment would delete the content of the Appendix B, Environmental Protection Plan (EPP) and modify License Conditions 2.C.(2) to delete that portion which refers to the EPP. Specifically, the requirements for non-radiological environmental monitoring have been completed. The radiological environmental monitoring requirements have been incorporated into Appendix A (the Technical Specifications). There would be no impact on the continued safety of the McGuire station by deleting Appendix B.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Deletion of the Environmental Protection Plan and modifying License Condition 2.C.(2) will have no impact on the probability or consequences of an accident previously evaluated because the changes will not have any impact upon the design or operation of any plant systems or components.

The proposed revision will not create the possibility of a new or different kind of accident from any previously evaluated because the revision is administrative in nature and will not change the types and amounts of effluent that will be released.

The proposed revision will not reduce a margin of safety because it is administrative in nature and will not effect the margin of safety as defined in the basis for any Technical Specifications.

Accordingly, this proposed changes does not involve a significant hazard.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

Duke Power Company, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: January 18, 1995.

Description of amendment request: The proposed amendments would relocate the requirements for the seismic instrumentation, meteorological instrumentation, and loose-part detection system from the Technical Specifications to the Selected Licensee Commitment (SCL) Manual. This will allow future changes to these controls to be performed under the provisions of 10 CFR 50.59. No changes are being made to the technical content of the affected Technical Specification pages.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Relocation of the affected TS sections to the SLC Manual will have no effect on the probability of any accident occurring. In addition, the consequences of an accident will not be impacted since the above instrumentation will continue to be utilized in the same manner as before. No impact on the plant response to accidents will be created.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms will be created as a result of relocating the affected TS requirements to the SLC Manual. Plant operation will not be affected by the proposed amendments and no new failure modes will be created.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. No impact upon any plant safety margins will be created. Relocation of the affected TS requirements to the SLC Manual is consistent with the content of the Westinghouse RSTS [Revised Standard Technical Specifications], as the NRC did not require technical specification controls for the affected instrumentation in the RSTS. The proposed amendments are consistent with the NRC philosophy of encouraging utilities to propose amendments that are consistent with the content of the RSTS.

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

Duke Power Company, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: January 18, 1995.

Description of amendment request: The amendments would revise Technical Specification Table 4.3–3 to allow the analog channel operational test interval for radiation monitoring instrumentation to be increased from monthly to quarterly. The proposed amendment changes would be consistent with the guidance in Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Decreasing the frequency of the radiation monitor analog channel operational test from monthly to quarterly will have no impact upon the probability or any accident, since the radiation monitors are not accident initiating equipment. Analysis of the previous test data * * * shows that no significant degradation of performance is to be expected by the decrease in frequency. Therefore, the requested amendments will have no adverse impact upon the consequences of any accident.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, the radiation monitors are not accident initiating equipment. No new failure modes can be created from an accident standpoint. The plant will not be operated in a different manner.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. Plant safety margins will be unaffected by the proposed changes. No safety equipment which is taken credit for in accident analyses will be affected by the requested amendments. The availability of the affected radiation monitors will be increased as a result of the proposed amendments because the monitors will not have to be made unavailable for testing as frequently. In addition, radiation monitor operating experience supports the proposed amendments. Finally, the proposed amendments are consistent with the NRC position and guidance set forth in NUREG–1366 and Generic Letter 93–05.

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: January 20, 1995.

Description of amendment request: The proposed amendments will relocate the operability requirements for the INCORE DETECTORS (TS 3/4.3.3.2) to the Updated Final Safety Analysis Report, and revise Linear Heat Rate surveillance 4.2.1.4, and Special Test Exceptions surveillances 4.10.2.2, 4.10.4.2 (Unit 2 only), and 4.10.5.2, accordingly.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature in that the specifications for operation and surveillance of the Incore Instrumentation (ICI) System will be relocated from the Technical Specifications to the Updated Final Safety Analysis Report for St. Lucie Unit 1 and Unit 2. Changes to the system will be controlled by 10 CFR 50.59, and the safety analysis report is required to be updated pursuant to 10 CFR 50.71(e). Relocation of these requirements to the UFSAR is consistent with the NRC ''Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors'' published in the **Federal Register** (58 FR 39132) dated July 22, 1993.

Incore instrumentation is not an accident initiator nor a part of the success path(s) which function to mitigate accidents evaluated in the plant safety analyses. The proposed technical specification change does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do the changes alter any assumptions or conditions in any of the plant accident analyses. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment to relocate the existing Technical Specification requirements for the Incore Instrumentation System to the Updated Final Safety Analysis Report will not change the physical plant or the modes of plant operation defined in the Facility License. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature in that operating and surveillance requirements for the Incore Instrumentation System will be relocated from the Technical Specifications to the Updated Final Safety Analysis Report for St. Lucie Unit 1 and Unit 2. The ICI system is not used to actuate safety-related equipment, provide interlocks, or otherwise perform automatic plant control functions. The system is used to monitor core power distribution parameters whose limits do involve a margin of safety; however, the ICI system itself makes no contribution to that margin of safety, and the power distribution limits will not be changed by the proposed amendment. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above discussion and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, DC 20036.

NRC Project Director: David B. Matthews.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: January 17, 1995.

Description of amendment request: The licensee proposes to revise the technical specifications to reference Topical Report NF-TR–95–01 as the documentation of the licensee's proficiency in performing certain reload design calculations once the NRC has evaluated and approved NR–TR–95–01.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The addition of the reference to FPL [Florida Power and Light Company] topical report which demonstrates FPL's ability to perform certain reload design calculations for Turkey Point Units 3 and 4 is administrative in nature and has no impact on the probability or consequences of any Design Bases Event (DBE) occurrences previously evaluated. The reload design calculations will be performed using methodologies and computer codes approved by the NRC and poses no increase in the probability or consequences of any accident previously evaluated.

The Core Operating Limits Report (COLR) parameters will be evaluated every cycle to ensure proper compliance with the Updated Final Safety Analysis Report (UFSAR). These limits will be evaluated in accordance with 10 CFR [Section] 50.59, which ensures that the reload will not involve an increase in the probability of occurrences or consequences of an accident previously evaluated. Title 10 CFR [Section] 50.59 (2) states that a proposed change involves an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. Consequently, since any change to the reload core design analysis must be evaluated relative to the more restrictive evaluation criterion of 10 CFR [Section] 50.59, then operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The addition of the reference to FPL topical report which demonstrates FPL's ability to perform certain reload design calculations for Turkey Point Units 3 and 4 is administrative in nature and has no impact, nor does it contribute in any way to the possibility of a new or different kind of accident from any accident previously evaluated. No new accident scenarios, failure mechanisms or limiting single failure events are introduced as a result of the proposed change.

The generation of the Axial Flux Difference, Rod Bank Insertion limits and K(Z) curve will be performed using NRC-approved methodology and are submitted to the NRC, as a revision to the COLR, to allow the NRC staff to trend. The Technical Specifications will continue to require operation within the core operating limits and appropriate actions will be taken if these limits are exceeded.

Title 10 CFR [Section] 50.59 permits a licensee to make changes in the facility as described in the safety analysis report without prior Commission approval, provided that the proposed changes does not involve an unreviewed safety question. 10 CFR [Section] 50.59 (2) states that a proposed change involves an unreviewed safety question (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created. Consequently, since any change to the reload core design analysis must be evaluated relative to the more restrictive evaluation criterion of 10 CFR [Section] 50.59, then operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The margin of safety is not affected by FPL performing the reload design calculations for Turkey Point Units 3 and 4. The supporting Technical Specification values are defined by the accident analyses which are performed to conservatively bound the operating conditions defined by the Technical Specifications. The development of the limits for future reloads will continue to conform to the methodology described in NRC approved documentation. In addition, each future reload will involve a 10 CFR [Section] 50.59

review to assure that operation of the units within the cycle specific limits will not involve a reduction in a margin of safety. 10 CFR [Section] 50.59 (2) states that a proposed change involves an unreviewed safety question (iii) if the margin of safety as defined in the basis for any technical specification is reduced. Consequently, since any change to the reload core design analysis must be evaluated relative to the more restrictive evaluation criterion of 10 CFR [Section] 50.59, then operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration. The NRC staff, however, considers that the licensee's statements relative to 10 CFR Section 50.59 evaluations to be performed in the future are not relevant to the proposed no significant hazards determination.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzer, P.C., 1615 L Street, NW., Washington, DC 20036.

NRC Project Director: David B. Matthews.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: October 28, 1994.

Description of amendment request: The proposed amendment revises the Duane Arnold Energy Center (DAEC) Operating License by deleting a condition of the license that requires a ''Plan for Integrating Scheduling of Plant Modifications for the Duane Arnold Energy Center'' (the Plan).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is provided below:

(1) The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. No physical changes to the facility will occur as a result of this amendment. Work activities will continue to receive the appropriate level of review in accordance with DAEC procedures and practices. The organizational structure that controls and manages these activities remains unchanged and will assure that activities are prioritized and performed in a manner consistent with plant safety. The proposed amendment removes an administrative burden that is no longer required.

(2) The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. No changes to the physical design and/or operation of the plant will occur as a result of this amendment. The processes by which activities are planned, prioritized, and controlled are not affected. The appropriate level of technical review and management oversight continue to be performed in accordance with existing procedures and practices to assure that activities are performed in a manner consistent with plant safety.

(3) The proposed amendment does not involve a significant reduction in a margin of safety. As stated earlier, no changes to the physical design and/or operation of any plant systems will occur as a result of this amendment. Work activities will continue to receive the appropriate technical review and management oversight to assure that activities are prioritized and performed in a manner consistent with plant safety. The amendment removes an administrative burden that is no longer required.

Based on the above, we have determined that the proposed amendment will not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis & Bouckins, 1800 M Street NW., Washington, DC 20036.

NRC Project Director: Leif J. Norrholm.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: January 24, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification 3.4.1, ''Leakage Rate,'' to reduce the allowable leakage rate of the reactor building from 2000 cubic feet per minute (cfm) to 1600 cfm.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Secondary containment and RBEVS [Reactor Building Emergency Ventilation System] are not initiators or precursors to an accident. Secondary containment provides a pressure boundary, with limited in-leakage, for the purpose of preventing a ground level unfiltered release of radioactivity. RBEVS responds to accidents involving release of radioactivity to the secondary containment by maintaining a negative pressure inside secondary containment and by providing an elevated release. Therefore, a change to the Reactor Building leakage rate cannot affect the probability of an accident previously evaluated.

Although the proposed change reduces the Reactor Building leakage rate from 2000 cfm to 1600 cfm consistent with system design, there is no effect on the radiological consequences of any previously analyzed accident since the radiological analysis does not assume exfiltration. Therefore, the Technical Specification change does not significantly increase the consequences of a previously evaluated accident.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to the Reactor Building leakage rate from 2000 cfm to 1600 cfm does not involve any accident precursors or initiators. During an accident involving a release of radioactivity to the secondary containment, the RBEVS would be operable and provide filtration of containment atmosphere prior to release to the environment. This change does not involve any physical modifications to the system, thus the system will operate as designed. Therefore, the proposed Technical Specification change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed change in Reactor Building in-leakage from 2000 cfm to 1600 cfm in Specification 3.4.1 and the associated basis is to be consistent with system design and reflect the leakage rate associated with approximately one building air volume change per day. The resulting accident analysis remains unchanged since the radiological analysis does not assume any exfiltration. Therefore, the proposed change will not involve a significant reduction in the margin of safety as defined in the basis for any Technical Specification.

Therefore, as determined by the above analysis, this proposed amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: Ledyard B. Marsh.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: February 1, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.6.13, ''Remote Shutdown Panels.'' TS 3.6.13 currently requires that if the valve controls or monitoring instrumentation on the Remote Shutdown Panels are inoperable, they must be restored to an operable status within 24 hours or the plant shall be shut down. The proposed change would require inoperable valve control functions be restored to an operable status within 30 days or the plant shall be shut down. The proposed change would also specify that required inoperable monitoring instrumentation functions be restored to an operable status within 30 days or that an alternate method of monitoring the parameter be established within 30 days and the required function be restored to an operable status within 90 days or the plant shall be shut down.

The proposed amendment would also make minor editorial changes to TS Table 3.6.13–1 so that the table entries would be consistent with the proposed revisions to TS 3.6.13.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The remote shutdown panel monitoring instruments and controls are not initiators or precursors to an accident. The remote shutdown panels provide the operator with sufficient monitoring instruments and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. Therefore, the proposed changes to Specification 3.6.13, ''Remote Shutdown Panels,'' cannot affect the probability of a previously evaluated accident.

The proposed changes, in part, require that one channel (on either panel) for each function be operable. This change could potentially avoid an unnecessary plant shutdown without affecting an operator's ability to cope with a control room evacuation. One channel of each function is adequate to assure a safe shutdown. The proposed changes would also allow 30 days to restore an inoperable function to an operable status. As indicated in the ITS [Improved Standard Technical Specifications], the allowed time of 30 days is acceptable based on operating experience and the low probability of an event that would require evacuation of the control room. With one or more monitoring instrument functions inoperable, the proposed change gives an operator an additional option. Specifically, the operator is allowed 30 days to establish an alternate method of monitoring the parameter and 90 days to restore the function to operable status. The use of an alternate method is acceptable since it will provide the operator with indication of the parameter of interest. The remote shutdown panels will not be required to be operable in hot shutdown because the plant is already subcritical and in a condition of reduced reactor coolant inventory energy. Because this Specification no longer applies to hot shutdown and to be consistent with the guidance provided in the ITS, Specification 3.6.13.d will require that the plant be brought to a hot shutdown condition (versus cold shutdown condition) in 12 hours. As indicated in the ITS, the 12-hour completion time is reasonable based on operating experience. The Bases Section to 3.6.13 and 4.6.13 was revised to be consistent with the proposed changes to the Specification. The Bases currently indicates that one remote shutdown panel is required to be operable. As explained above, one channel of each required function is required to maintain remote shutdown operability. In summary, the proposed changes will not affect the ability of the Remote Shutdown System to provide the operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. Therefore, the consequences of an event requiring a control room evacuation will not significantly increase.

Editorial changes were made to Table 3.6.13–1 to be consistent with the changes made to the Specification. Specifically, the word ''INSTRUMENT'' was changed to ''FUNCTION'' and the words ''PANEL MONITORING'' were changed to the words ''PANELS FUNCTIONS.'' These changes make it clear that one channel of each function, on either panel is acceptable to maintain operability. The emergency condenser condensate return valve control and motor-operated steam supply valves control were relocated from Specification 3.6.13.b to Table 3.6.13–1 to be consistent with the proposed changes.

Based on the above, the consequences of an accident previously evaluated are not significantly increased.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes do not introduce any new accident precursors and do not involve any alterations to plant configurations which could initiate a new or different kind of accident. The proposed changes require that one channel of each function be operable to assure the remote shutdown panels can meet their intended function. No changes have been made which will affect the operation of the remote shutdown panels in a way which would create a new or different kind of accident. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed changes will not affect the ability of the Remote Shutdown System to provide the operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. The ability to respond to a control room evacuation is maintained with one channel operable for each required function. The allowed outage time of 30 days is acceptable based on operating experience and the low probability of an event requiring control room evacuation. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: Ledyard B. Marsh.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: January 10, 1995.

Description of amendment request: The proposed amendment request would revise Technical Specifications by deleting the power range, neutron flux, high negative rate trip from Tables 2.2–1, 3.3–1, and 4.3–1, and delete the associated Bases Section 2.0.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

* * * The proposed changes would not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The deletion of the power range, neutron flux, high negative rate trip will not adversely affect plant operations. As has been presented and accepted by the NRC Staff in previous docketed correspondence, the dropped RCCA [rod cluster control assembly] accident analysis does not rely on this trip to safely shut down the plant. The safety analysis of the plant is unaffected by the proposed changes. Since the safety analysis is unaffected, the calculated radiologicalreleases associated with the analysis are not affected. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The reactor trip system is used to mitigate accidents. There have been instances, during calibration of these units, where a single channel has generated a trip signal. Leaving this in place when it is not necessary could, therefore, cause a reactor trip. The deletion of one trip function will, therefore, slightly decease, not increase, this probability.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The reactor trip system is used to mitigate accidents, and the only way that it can initiate an event is by causing the reactor to trip when it is unnecessary. This possibility of the generation of a false trip signal has already been evaluated in the safety analysis. This modification will physically remove or disable the power range, neutron flux trip and will therefore decrease the possibility for the generation of a false trip signal. Therefore, the proposed change cannot create a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed change which deletes the power range, neutron flux, high negative rate trip will have no impact on the margin of safety. The current safety analysis for Millstone Unit No. 3 does not credit this trip for any events; therefore, removal of this trip from the technical specifications will not affect the margin of safety for any analyzed events.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: January 23, 1995.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) by 1) adding a new Section 3/4.5.5 which provides a limiting condition for operation, an action statement, a surveillance requirement, and a corresponding bases section, for the trisodium phosphate (TSP) baskets which will be installed in the next refueling outage; 2) deleting Section 3/4.6.2.3 and Bases 3/4.6.2.3 related to the spray additive system which are no longer needed since the chemical addition tank is being abandoned; and 3) updating Index Pages viii, ix, and xiv to reflect the above changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

* * * The proposed changes do not involve an SHC because the changes would not:

1. Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The plant change affects the chemical composition of the QSS [quench spray system] flow and the method of sump pH control, which are important for containment heat removal/pressure mitigation (MSLB and LOCA) [main steamline break and loss-of coolant accident] and fission product removal (LOCA). However, this change does not affect the probability of occurrence of these accidents. Since the TSP baskets are passive devices located inside the containment, they cannot initiate a transient or affect the probability of occurrence of any previously evaluated accident.

The design change will not adversely affect the radiological doses for the DBA [design basis accident] LOCA at the Exclusion Area Boundary, Low Population Zone, Millstone Unit No. 3 Control Room, Millstone Unit No. 2 Control Room, and the Millstone Technical Support Center. Also, the change will not adversely affect the calculated peak clad temperature (PCT) for the DBA LOCA.

2. Create the Possibility of a New or Different Kind of Accident from any Previously Analyzed.

The change does not create a malfunction that is different from those previously evaluated. The TSP baskets are passive devices that have minimal impact on any other systems except through water chemistry. The change in water chemistry does not adversely affect any safety systems. The installation of the TSP baskets and the abandonment of the CAT [chemical addition tank] will not change the probability of a malfunction of safety-related equipment.

Potential malfunctions relating to the TSP powder, the 12 baskets which hold the TSP powder, the QSS and other systems, and equipment credited in the safety analysis were evaluated and determined not to be adversely affected by the change. Additionally, the transient pH behavior of the spray flow will not adversely affect metals, coatings and elastomers in the containment, and the performance of associated safety functions is not affected.

Finally, the change in the chemical composition of the QSS solution will not affect the operability of this system or its ability for containment heat removal and pressure mitigation.

3. Involve a Significant Reduction in the Margin of Safety.

The design changes do not adversely affect the ability of the QSS to perform the function of containment heat removal, pressure mitigation and fission product (iodine) retention. The design changes do not adversely affect any equipment credited in the safety analysis. Also, the design changes to not increase the calculated peak clad temperature (PCT) or the offsite doses due to the design basis LOCA. Therefore, there is no impact on the margin of safety as specified in the technical specifications.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: January 24, 1995.

Description of amendment request: The amendment request would revise the Technical Specification Section 3.2.3.1.a and Table 2.2–1 to decrease the acceptance criterion for measured reactor coolant system (RCS) flow rate from 387,480 gallons per minute (gpm) to 371,920 gpm.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

* * * The proposed changes do not involve an SHC because the changes would not:

1. Involve a Significant Increase in the Probability or Consequence of an Accident Previously Evaluated.

An evaluation of the 4% decrease in the RCS total flow rate limit has shown that the change does not significantly impact the design basis analyses. Therefore, the change will not increase the consequences of an accident previously evaluated.

There are no actual plant changes that will result from this technical specification change. Instead, the technical specification requirement for minimum total RCS flow rate is being changed to provide operational benefit without compromising safety. Since there are no plant changes, there is no effect on the probability of occurrence of previously evaluated accidents.

The change will have a negligible impact on the small break loss of coolant accident (LOCA) and large break LOCA analyses. The PCT [peak cladding temperature] acceptance criteria will continue to be met with the assumption of a 4% reduction in RCS flow rate.

For the steam generator tube rupture event, both the FSAR [Final Safety Analysis Report] offsite dose analysis and the margin of steam generator (SG) overfill were evaluated. It was determined that the 4% reduction in RCS flow rate will not adversely affect the offsite doses or the margin to SG overfill and, therefore, the FSAR conclusions remain unchanged.

In the evaluation of non-LOCA transients, the DNB [departure from nucleate boiling] is the most affected parameter due to a change in flow rate. It was concluded that the 4% reduction in RCS flow was acceptable and there was margin to the DNB limit.

It is concluded that there is sufficient margin to the system pressure, PCT and DNB limits to offset the effect of the 4% flow rate decrease and the calculated radiological releases associated with the analysis are not affected. Therefore, there is no effect on the consequences of previously evaluated accidents.

2. Create the Possibility of a New or Different Kind of Accident from any Previously Analyzed.

The low loop flow trip setpoint specified in Technical Specification Table 2.2–1 is set as a fraction of total flow. The flow fraction is not being changed and no hardware changes are required due to the reduction in minimum flow. Also, the reduction in minimum flow will not change the operation of any plant equipment and it does not modify plant operation.

Therefore, the reduction in minimum flow does not introduce any new failure modes or malfunctions and it does not create the potential for a new unanalyzed accident.

3. Involve a Significant Reduction in the Margin of Safety.

The proposed 4% decrease in the technical specification limit for total RCS flow rate will not adversely affect the results of the FSAR accident analysis, and it is concluded that this change is safe. The change does not adversely affect any equipment credited in the safety analysis, and it does not affect the probability of occurrence of any plant accident. Also, the change has a negligible impact on the PCT, and it does not increase the offsite doses or decrease the DNB below its acceptance limit.

Therefore, the change does not have any significant impact on the protective boundaries, and there is no reduction in the margin of safety as specified in the technical specifications.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: January 9, 1995.

Description of amendment request: The proposed amendment to the technical specifications (TSs) would delete requirements for the toxic gas monitoring system (TGMS) as contained in TS 2.22 and TS 3.1, Table 3–3, item 29.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The previously evaluated accidents affected by this change are the on-site and off-site toxic chemical releases. These events have been re-evaluated for this proposed change and have been shown to meet the applicable regulatory screening criteria. The deterministic analyses performed show that the guidelines of Regulatory Guide 1.78 for control room habitability are met for on-site and most off-site chemicals. On-site chemical sources originally present when the toxic gas monitoring system was installed have been removed from site or determined not to exceed the deterministic analysis screening requirements. For those off-site chemical releases which did not meet the deterministic screening criteria a probabilistic analysis was performed. The probabilistic analysis performed in support of this proposed change shows that the probability of an off-site chemical release leading to 10 CFR 100 consequences is orders of magnitude less than the SRP [Standard Review Plan] 2.2.3 guidelines. These results show that there is no significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

Only events involving chemicals for which the TGMS provides an automatic detection/isolation function are affected by this change. As stated above, the potential events involving these chemicals have been re-evaluated using the appropriate regulatory guidance and shown to satisfy either the deterministic screening criteria of RG [Regulatory Guide] 1.78, or to be probabilistically insignificant compared to the guidelines of SRP Section 2.2.3. These results show that the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

(3) The proposed changes do not involve a significant reduction in a margin of safety.

The margin of safety is defined by the regulatory basis for the existing TGMS, namely NUREG–0737, Item III.D.3.4. The analysis provided to support this proposed change follows the regulatory guidelines of RG 1.78 and SRP Section 2.2.3, as specified in NUREG–0737, Item III.D.3.4. The analysis shows that the applicable regulatory criteria are met and the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1875 Connecticut

Avenue, NW., Washington, DC 20009–5728.

NRC Project Director: Theodore R. Quay.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: February 6, 1995 (Reference LAR 95–01).

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2, to change TS 3/4.9.14.1, ''Spent Fuel Assembly Storage,'' TS 3/4.9.14.2, ''Spent Fuel Pool Boron Concentration,'' TS 5.3.1, ''Reactor Core—Fuel Assemblies,'' and TS 5.6.1, ''Fuel Storage—Criticality,'' and add new TS 3/4.9.14.3, ''Spent Fuel Assembly Storage—Spent Fuel Pool Region 1.'' The specific TS changes proposed are as follows:

(1) The proposed changes to TS 3/4.9.14 are:

(a) TS 3.9.14.1 and Figure 3.9–2 would be revised to allow the storage of spent fuel assemblies with initial enrichments up to 5.0 weight percent uranium-235 (U–235) in Region 2 of the spent fuel pool (SFP). Fuel pellet diameter would be considered in combination with initial enrichment and cumulative burnup.

(b) Editorial corrections to the titles of TS 3/4.9.14.1 and 3/4.9.14.2 would be made for consistency with the TS format.

(2) New TS 3/4.9.14.3 would be added. The new TS would include:

(a) Requirements for acceptable fuel storage in Region 1 of the SFP.

(b) An action statement, similar to that for TS 3.9.14.1, requiring suspension of all fuel movement and crane operations except to move the noncomplying fuel assemblies into an acceptable pattern. The action statement also requires verification of SFP boron concentration at least once per 8 hours.

(c) A requirement, similar to that for TS 4.9.14.1, for an evaluation that considers enrichment, boron content, and cumulative burnup of each fuel assembly before storage in Region 1 of the SFP.

(d) New Figure 3.9–3 for use in determining the acceptability of storing fuel in Region 1 of the SFP.

(3) The proposed changes to TS 5.3.1 are:

(a) The number of fuel rods in each fuel assembly, nominal length of each fuel rod, and maximum fuel enrichment would be removed.

(b) The current allowance for fuel rod substitutions as justified by analysis would be clarified to specify that the analysis be performed using NRC staff-approved methods.

(c) An allowance to use a limited number of lead test assemblies in nonlimiting core locations would be added.

(d) The current specification requiring Zircaloy-4 fuel cladding would be changed to allow Zircaloy-4 or ZIRLO cladding.

(4) The proposed changes to TS 5.6 are:

(a) TS 5.6.1.1 would be renumbered TS 5.6.1 and the word ''borated'' would be replaced with ''unborated.''

(b) A new requirement would be added to specify the maximum fuel enrichment allowed to be stored in the fuel racks.

(c) TS 5.6.1.2 would be deleted.

(5) The associated Bases would also be appropriately revised.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

a. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Analyses were performed to verify that an increase in enrichment of the fuel from 4.5 weight percent U–235 to 5.0 weight percent U–235 would not result in an inadvertent criticality event in the new fuel storage racks or the SFP. The analyses indicate that for the new fuel racks, the k_{eff} will remain below 0.95 if flooded with non-borated water, and below 0.98 if flooded with optimum-density aqueous foam. The analyses indicate that for the spent fuel racks, assuming credit for soluble boron in accident scenarios, the k_{eff} will remain below 0.95 as required.

The increase in the fuel enrichment from 4.5 weight percent U–235 to 5.0 weight percent U–235 does not change any of the external dimensional characteristics of the fuel element, the fuel storage racks, or the SFP itself. The accidents originally evaluated considered those events that could lead to fuel damage and release of radioactive material primarily from mechanical means, such as physical impact on the fuel or the SFP. Because the physical design and methods of operation are the same as previously evaluated, there is no change in the probability of occurrence of such events.

The maximum spent fuel gap activity and the resulting offsite dose consequences after a postulated fuel handling accident are primarily dependent on fuel burnup, and are not significantly affected by an increase in fuel enrichment. For up to 5.0 weight percent U–235 and 60,000 MWD/MTU burnup, NUREG/CR–5009 indicates that fuel handling accident offsite doses could increase by a factor of 1.2, which indicates that doses would still remain within 10 CFR Part 100 limits.

The Generic Letter 90–02 Supplement 1 change to TS 5.3.1 clarifies the requirements associated with fuel reconstitution. It does not change the methodology that would be used to reconstitute fuel.

The use of ZIRLO cladding will not increase the probability or consequences of an accident, since it has improved mechanical properties such as a lower corrosion rate and reduced radiation-induced growth.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

b. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The physical and mechanical parameters associated with the fuel assemblies and spent fuel racks are the same as previously evaluated. Therefore, any malfunctions related to the physical aspects of fuel storage are the same as previously evaluated.

The conditions for fuel storage in the proposed new TS 3.9.14.3 provide new criteria for locations where a fuel assembly could be incorrectly placed. However, the incorrect placement of a fuel assembly has been analyzed, and would not cause an inadvertent criticality or any other accident.

The change to 5.0 weight percent U–235 does not result in physical alterations or changes to the operation of the plant, or change the method by which any safety-related system performs its function. The use of ZIRLO cladding does not result in a significant change to the plant.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

c. Does the change involve a significant reduction in a margin of safety?

The acceptance criteria of a k_{eff} of 0.95 (or 0.98 for the new fuel rack optimum moderation accident) provides the margin to criticality. Analyses were performed that conclude that the proposed changes to allow up to 5.0 weight percent U–235 in the new and spent fuel racks meet the acceptance criteria. The use of ZIRLO cladding will not reduce the protection of the public health or safety, as indicated in the NRC's revisions to 10 CFR 50.44 and 50.46 (57 FR 39355).

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps

Department, San Luis Obispo, California 93407.

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: Theodore R. Quay.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: November 23, 1994.

Description of amendment request: The proposed amendment would revise the Technical Specifications Section VI, ''Waste Disposal Systems,'' regarding radioactive effluent limitations and the conditions for automatically pumping the contents of the reactor caisson sump to the outfall canal.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed revisions to the HBPP Technical Specifications remove the ambiguity in the guidelines for directing caisson sump discharges to the outfall canal. Additionally, the proposed revisions will modify Section VI to be consistent with the guidance provided by NRC Draft Generic Letter for 10 CFR 20 Modification to Technical Specifications (58 FR 68171, dated December 23, 1993). These changes in effluent limits are not related to the probability or consequences of an accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed revisions to the HBPP Technical Specifications are administrative in nature and do not change the method by which any safety-related system performs its function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed revisions to the HBPP Technical Specifications do not affect the margin of safety associated with parameters for any accident analysis.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensee and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Humboldt County Library, 636 F Street, Eureka, California 95501.

Attorney for licensee: Christopher J. Warner, Esquire, Pacific Gas & Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: Seymour H. Weiss.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: November 23, 1994.

Description of amendment request: The proposed amendment would revise the Technical Specifications Section VII.C, Plant Staff, to decrease the minimum staff requirements for the shift operating organization from five to two persons.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The probability or consequences of an accident previously evaluated will not be affected by the change in plant staffing. The plant staff manning requirements for the shift operating organization are being reduced to reflect the condition of the plant in a SAFSTOR mode. Previously evaluated accidents do not require operator actions to mitigate or reduce the consequences of occurrence. Consequently, the change will not affect the probability or consequences of an accident occurring.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed revisions to the HBPP Technical Specifications are administrative in nature. Further, there would not be any change in equipment or system function or operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed revisions to the HBPP Technical Specifications do not affect the margin of safety of any accident analysis since they do not affect the parameters for any accident analysis, and they have no effect on the current operating methodologies or actions that govern plant performance.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensee and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Humboldt County Library, 636 F Street, Eureka, California 95501.

Attorney for licensee: Christopher J. Warner, Esquire, Pacific Gas & Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: Seymour H. Weiss.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: January 13, 1995.

Description of amendment request: The proposed changes revise Tables 3.7.1 and 3.7.4 to reflect a reduction in the number of primary containment power operated outboard valves for the Traversing Incore Probe (TIP) probes, and a redesignation of the containment penetration numbers for the TIP ball, shear, and check valves. The proposed changes are a result of PBAPS Modification P00068.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The TIP system does not serve as an initiator or contributor to any accidents previously evaluated. The system provides a means of calibrating the Local Power Range Monitors and supports thermal limit calculations. The new system performs the same function as the old one. It will provide improved reliability and added redundancy by allowing a complete flux mapping if a detector or drive failure were to occur.

Installation of Modification P00068 and its operation will not degrade any active or passive equipment that responds to an accident. These changes do not decrease the effectiveness of equipment relied upon to mitigate the previously evaluated accidents.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The modification is considered an enhancement to the TIP system and does not serve as an initiator or contributor to any of the accidents previously evaluated. The proposed changes do not introduce a new mode of plant operation. The new system, like the old one, is designed to keep the ball valves closed upon reset of the Primary Containment Isolation System (PCIS) logic. The new TIP control console will respond to a PCIS isolation signal in the same manner as the old system.

Implementation of the proposed changes will not affect the design function or configuration of any component or introduce any new operating scenarios or failure modes or accident initiation.

Modification P00068 will not impair or prevent safety systems from performing their safety function. It will not make any changes to the design function of the TIP system. The classification of the TIP ball and shear valves and their control circuitry will not change as a result of this modification.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The TIP system does not serve as an initiator or contributor to any accidents evaluated in the SAR [safety analysis report]. Modification P00068 is considered an enhancement to the existing TIP system and does not change its design function. The reduction in the number of containment penetrations from five to three does not represent a reduction in a margin of safety because of additional indexers in the new system. The proposed changes do not adversely affect the assumptions or sequence of events used in any accident analysis.

Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Attorney for Licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101.

NRC Project Director: John F. Stolz.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: June 13, 1994.

Description of amendment request: The proposed change would remove license condition 2.E from the Facility Operating License. License Condition 2.E incorporated the requirements of U.S. Department of Interior publication ''Environmental Criteria for Electric Transmission Systems''—1970, which applies to the construction cleanup, restoration, and maintenance of transmission lines.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will remove a license condition unrelated to nuclear safety. License condition 2.E incorporated into the Operating License the requirements of U.S. Department of Interior publication ''Environmental Criteria for Electric Transmission Systems''—1970. The goal of this standard is to ''safeguard aesthetic and environmental values within the constraints imposed by the current state of high-voltage transmission technology.'' License condition 2.E addresses the preservation of the environment and natural resources. Removing this condition from the Facility Operating License has no bearing on plant safety or the health and safety of the public considering its non-nuclear nature. The transmission line right-of-ways maintained by the [Power] Authority [of the State of New York] are subject to regulation by other State and Federal Agencies. Removal of this license condition will not affect operation of safety related structures, systems or components nor affect the quality assurance program at the FitzPatrick plant. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated.

License condition 2.E of the James A. FitzPatrick Plant Operating License applies to the construction cleanup, restoration, and maintenance of transmission lines. The Authority's transmission lines are managed under guidelines based on the ''Generic Transmission Line Right-of-Way Management'' plan requirements. The requirements imposed by the plan on the FitzPatrick transmission line right-of-ways exceed those of the U.S. Department of Interior publication referenced in license condition 2.E in both scope and details. Therefore, implementing the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) involve a significant reduction in a margin of safety.

License condition 2.E of the James A. FitzPatrick Plant Operating License applies to the construction cleanup, restoration, and maintenance of transmission lines. The requirements imposed by this license condition are unrelated to nuclear safety. Continued operation of the plant without Condition 2.E does not involve a significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: December 16, 1994; supplemented February 10, 1995 (TS 94–07).

Description of amendment request: The proposed change would reduce the maximum allowed power levels and more clearly specify the plant conditions allowed by the technical specifications for operation with one or more main steam safety valves inoperable. In addition, the Bases would be revised to reflect these changes and incorporate the revised methodology used to establish the neutron flux setpoints.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of

Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

This change reduces the power level at which the reactor may be operated with one or more main steam safety valves (MSSVs) inoperable, to ensure that the secondary system is not overpressurized during the most severe pressurization transient of the secondary side. Additionally, this change will combine the TS action statements for 3- and 4-loop operation with one or more MSSVs inoperable, revise the mode requirements and times of Action Statement 3.7.1.1.a, and correct a reference in the bases section to Table 3.7–1. Reduction of the high neutron flux (HNF) trip setpoint will continue to be used as the means to ensure that the required reactor power level reductions are met. Mode 3 will be limited to application when the reactor trip breakers (RTB) are closed. Lack of NIS trip setpoint adjustments with the RTB open has no effect on the accident analysis. There is no change to the function of the MSSVs by the proposed change. This change will not alter any accident analysis assumptions or results for SQN. The proposed change will reduce the amount of relief capacity required to mitigate the consequences of the transient by reducing the total amount of energy in the primary system. Therefore, this change will not increase the probability of an accident.

This change is consistent with current SQN accident analysis assumptions for the MSSVs and does not change the containment response for any design basis event. Therefore, no change in the mitigation of an accident will result from this proposed change and no change will occur in the consequences of any accident currently analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

Inadvertent opening of a MSSV is currently analyzed as an initiating event for accidental depressurization of the main steam system. The proposed change does not alter the valves or any other plant component. The valves will continue to perform as analyzed in current accident analyses. The proposed change will not create the possibility for any new or different kind of accident.

By retaining the use of the HNF trip setpoint reduction, no change is being proposed in the methodology used to ensure that power reductions are carried out; therefore, this will not create the possibility of placing the plant into any new unanalyzed condition. Not adjusting the Nuclear Instrumentation System trip setpoint with the RTBs open will not create an accident. The existing accident analysis is still bounding.

Combining the separate action statements for 3- and 4-loop operation into a single action does not create the possibility for a new or different kind of accident. Operation with 4 loops will continue to be required in Modes 1 and 2 by TS 3.4.1.1.

Operation with less than 4 loops will continue to be governed by TS 3.4.1.2 in Mode 3 and TS 3.4.1.3 in Mode 4. This change will not place the plant in a configuration not currently bounded by existing accident analysis.

Revising the mode requirements and their associated times, consistent with the requirements in NUREG–1431, will continue to ensure that if the unit is unable to comply with the limiting condition for operation, the unit will begin an orderly shutdown until a mode is reached where the specification is not applicable.

3. Involve a significant reduction in a margin of safety.

The proposed change reduces the total energy of the reactor coolant system that will ensure the ability of the MSSVs to perform their intended function as assumed in current accident analyses. This change has been evaluated on a generic basis for Westinghouse Electric Corporation designed 4-loop nuclear steam supply systems. SQN plant specific features have been evaluated including power limit calculations and the interaction of the reactor protection system trip time delay and the anticipated transient without scram mitigating system actuation circuitry. Correcting this nonconservatism restores the margin of safety to what was originally envisioned. Therefore, the margin of safety assumed in the accident analysis is not reduced by this change.

Combining the separate action statements for 3- and 4-loop operation into a single action has no effect on the margin of safety for 4-loop operation with one or more MSSVs inoperable. Under the revised TS, 3-loop operation with one or more MSSVs inoperable would only be allowed in Mode 3, and 4-loop operation will be required in Modes 1 and 2 in accordance with current TSs 3.4.1.1 and 3.4.1.2.

Revising the mode requirements and their associated times, consistent with the requirements in NUREG–1431, will not reduce the safety margin since the new requirements will continue to place the unit in a mode where the TS is no longer applicable. The new completion times for mode changes are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

The margin of safety is unaffected by modifying the limits of Mode 3 applicability to require the RTBs to be closed as the intended safety function is already completed when they are open.

The NRC has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: December 9, 1994, and January 27, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement 4.6.1.2.a and its associated Bases. The changes would defer the next scheduled containment integrated leak rate test (CILRT) for one outage, from Refuel 7 (March 1995) to Refuel 8 (scheduled for September 1996).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

The Callaway CILRT history provides substantial justification for the proposed test schedule. Three CILRTs have been performed over a seven year period with successful results. The tests indicate that Callaway has a low leakage containment. There are no structural mechanisms which would adversely affect the structural capability of the containment and that would be a factor in extending the CILRT schedule by one refueling outage.

A risk impact assessment was performed, and a determination was made that there is insignificant risk impact as a result of changing the CILRT schedule. Containment leak rate testing is not an initiator of any accident, the proposed interval extension does not affect reactor operations or the accident analysis, and has no radiological consequences. There will be no changes to 10 CFR 100 dose limits or the control room dose limits. Extending the test interval will not, by itself, increase the probability of a malfunction of equipment important to safety. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated in the Safety Analysis Report.

There are no design changes being made that would create a new type of accident or malfunction. The proposed change will not alter the plant or the manner in which it is operated. The change revises the schedule for performing the periodic CILRT. The purpose of the test is to provide periodic verification of the leaktight integrity of the primary reactor containment, and systems and components which penetrate containment. The tests assure that leakage through containment and systems and components penetrating containment will not exceed the allowable leakage rate values associated with conditions resulting from an accident. The change in schedule for performing the CILRT will not adversely affect the containment integrity in the event of an accident. Therefore, the proposed change will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed change to the schedule for performing the periodic Type A test does not reduce the margin of safety assumed in the accident analysis for any release of radioactive materials or reduce any margin of safety preserved by the technical specifications. The methodology, acceptance criteria, and the technical specification leakage limits for the performance of the Type A tests will not change. The Type A tests will continue to be performed in accordance with 10 CFR 50, Appendix J and the Callaway Technical Specifications. Therefore, the proposed change will not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, DC 20037.

NRC Project Director: Leif J. Norrholm.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Power Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: January 24, 1995.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Section 15.6.5, ''Review and Audit,'' and TS Section 15.7.8, ''Administrative Controls.'' The quality assurance audit frequencies would be removed, the section on emergency plan reviews would be removed, and the period for radioactive effluent reporting would be increased to annual. In addition, the references to ''Semiannual Monitoring Report'' would be changed to ''Annual Monitoring Report'' throughout TS Section 15.7.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

In accordance with the requirements of 10 CFR 50.91(a), Wisconsin Electric Power Company (Licensee) has evaluated the proposed changes against the standards of 10 CFR 50.92 and has determined that the operation of Point Beach Nuclear Plant, Units 1 and 2, in accordance with the proposed amendments, does not present a significant hazards consideration.

A proposed facility operating license amendment does not present a significant hazards consideration if operation of the facility in accordance with the proposed amendment will not:

1. Create a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will not create a significant reduction in a margin of safety.

The proposed changes are administrative in nature. There is no physical change to the facility, its systems, or its operation. Since the changes will allow more flexibility in assigning resources to work on poor or weak performance areas, the plant safety will be enhanced. Operation of PBNP in accordance with the proposed amendments cannot create an increase in the probability or consequences of an accident previously evaluated, create a new or different kind of accident, or result in a significant reduction in a margin of safety. Therefore, the proposed changes do not present a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Leif J. Norrholm.

Previously Published Notices of Consideration of Issuance of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Duke Power Company, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of amendment request: October 18, 1994.

Description of amendment request: The proposed amendment would change Technical Specification 3.6.1.2 to defer the next scheduled containment integrated leak rate test at Catawba Unit 1 for one outage, from the end-of-cycle (EOC) 8 refueling outage (scheduled for February 1995) to EOC 9 (scheduled for June 1996).

Date of publication of individual notice in **Federal Register**: February 6, 1995 (60 FR 7073).

Expiration date of individual notice: March 8, 1995.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina.

Duke Power Company, et al., Docket No. 50–413 Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of amendment request: November 29, 1994, as supplemented January 12 and 27, 1995.

Description of amendment request: The proposed amendment requested renewal for Catawba Unit 1 Cycle 9 operation of the steam generator tube inspection bobbin probe voltage-based interim plugging criteria that had been previously approved for Cycle 8.

Date of publication of individual notice in **Federal Register**: February 9, 1995 (60 FR 7801).

Expiration date of individual notice: March 13, 1995.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina.

Georgia Power Company, et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia,

Date of amendment request: January 20, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification 6.4.1.2 to provide a more accurate description of the Plant Review Board composition.

Date of publication of individual notice in **Federal Register**: February 6, 1995 (60 FR 7077).

Expiration date of individual notice: March 8, 1995.

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: November 7, 1994, as supplemented by letters dated December 20, 1994, and January 23, 1995.

Brief description of amendment request: The proposed amendments would change the number of diesel generators (emergency power supply) required to be operable during Mode 6 with greater than or equal to 23 feet of water above the reactor vessel flange, from two to one. The amendments would also allow limited substitution of an alternate onsite emergency power source for one of the two required diesel generators, in Mode 5 and in Mode 6 with less than 23 feet of water. In addition, changes to certain system specifications that are affected by the changes for the emergency power supply were also proposed.

Date of individual notice in **Federal Register**: January 30, 1995 (60 FR 5739).

Expiration date of individual notice: March 1, 1995.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50–461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: January 27, 1995.

Brief description of amendment request: The amendment modifies the technical specifications (TSs) by eliminating selected response time testing as described in the BWROG topical report NEDO–32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements.'' The affected TSs are TS 3.3.1.1, ''Reactor Protection System (RPS) Instrumentation,'' TS 3.3.5.1, ''Emergency Core Cooling System (ECCS) Instrumentation,'' TS 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' and TS 3.5.1, ''ECCS—Operating.''

Date of publication of individual notice in **Federal Register**: February 3, 1995 (60 FR 6739).

Expiration date of individual notice: March 6, 1995.

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: October 31, 1994, as supplemented by letter dated December 28, 1994.

Brief description of amendments: The amendments revise the refueling machine overload cutoff limit from less than or equal to 1556 pounds to less than or equal to 1600 pounds. The change was requested because design and fabrication improvements have increased the weight of the fuel assembly.

Date of issuance: February 9, 1995.

Effective date: February 9, 1995, to be implemented within 45 days of the date of issuance.

Amendment Nos.: 89, 76, and 60.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 6, 1995 (60 FR 2160). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: November 20, 1992, as supplemented by letters dated October 22, 1993, and November 30, 1994.

Brief description of amendments: The amendments would increase the allowable out-of-service time for the core operating limit supervisory system (COLSS) from 1 hour to 4 hours before the more restrictive limits based on the core protection calculators (CPCs) must be applied.

Date of issuance: February 14, 1995.

Effective date: February 14, 1995.

Amendment Nos.: 90, 77, and 61.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 6, 1993 (58 FR 591) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 14, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: September 6, 1994.

Brief description of amendment: The amendment would remove Technical Specification Section 4.5.H.4 which requires the testing and calibration of pressure switches in certain emergency core cooling system lines.

Date of issuance: February 2, 1995.

Effective date: February 2, 1995.

Amendment No.: 157.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 26, 1994 (59 FR 53838). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1994 (59 FR 53838).

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: March 25, 1994, as supplemented on July 29, 1994, and August 24, 1994.

Brief Description of amendments: The amendments change the Technical Specifications to correct several typographical errors, to incorporate material implicitly contained in a footnote to an applicability statement, to provide detailed labels for items listed in a table, to correct the citation of references, and to remove references to the Rod Sequence Control System that should have been included in a previous change.

Date of issuance: February 1, 1995.

Effective date: February 1, 1995.

Amendment Nos.: 174 and 205.

Facility Operating License Nos. DPR–71 and DPR–62. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27050). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 1, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: December 12, 1994.

Brief description of amendment: The amendment revises the containment spray (CS) nozzle surveillance interval from 5 to 10 years.

Date of issuance: February 10, 1995.

Effective date: February 10, 1995.

Amendment No.: 157.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: January 4, 1995 (60 FR 497).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Hartsville Memorial Library, 147 West College, Hartsville, South Carolina 29550.

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of application for amendments: June 24, 1994.

Brief description of amendments: The amendments revise the Technical Specifications by deleting the containment recirculation sump level from Accident Monitoring Instrumentation Tables 3.8.9–1 and 4.8.9–1.

Date of issuance: February 9, 1995.

Effective date: February 9, 1995.

Amendment Nos.: 160 and 148.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37066).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut, and Northeast Nuclear Energy Company, Docket Nos. 50–245, 50–336, and 50–423, Millstone Nuclear Power Station, Units 1, 2, and 3, New London County, Connecticut

Date of application for amendments: June 30, 1994, as supplemented November 18, 1994, and January 12, 1995.

Brief description of amendments: The amendments modify the Administrative Controls Section of the Technical Specifications by replacing the present Nuclear Review Board (NRB) for the Haddam Neck Plant, and the NRB and site Nuclear Review Board for Millstone Station with a Nuclear Safety Assessment Board which will serve Millstone Units 1, 2, and 3, and Haddam Neck.

Date of issuance: February 14, 1995.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 181, 79, 184, 104.

Facility Operating License Nos. DPR–61, DPR–21, DPR–65 AND NPF–49. Amendments revised the Technical Specifications.

The November 18, 1994, and January 12, 1995, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

Date of initial notice in **Federal Register**: August 31, 1994 (59 FR 45021).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Russell Library, 123 Broad Street, Middletown, CT 06457, for the Haddam Neck Plant, and Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360, for Millstone 1, 2, and 3.

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: July 29, 1994, as supplemented in a letter dated December 13, 1994.

Brief description of amendment: This amendment revises Technical Specifications (TSs) 3/4.4.5 and 3.4.6.2 including associated Bases 3/4.4.5 and 3/4.4.6.2 to allow the implementation of steam generator tube interim plugging criteria (IPC) for the tube support plate elevations during operating cycle 11. The current TSs require that tubes with imperfections exceeding 40 percent of the nominal tube wall thickness be removed from service. The IPC will allow a test voltage-based criterion of 1.0 volts as determined by a bobbin probe inspection of the tubes. Voltages greater than 1.0 volt will be further examined using a pancake coil probe. Tubes showing flaw indications with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

Date of issuance: February 3, 1995.

Effective date: February 3, 1995.

Amendment No: 184.

Facility Operating License No. DPR–66. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 17, 1994 (59 FR 42337). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: November 8, 1994.

Brief description of amendment: The amendment revised the technical specification section that describes the frequency for performing the containment integrated leak rate tests.

Date of issuance: February 6, 1995.

Effective date: February 6, 1995.

Amendment No.: 175.

Facility Operating License No. DPR–51. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 4, 1995, (60 FR 502). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: July 25, 1994.

Brief description of amendment: This amendment will upgrade Technical Specification 3/4.7.1.6 for the Main Feedwater Line Isolation Valves to be consistent with NUREG–1432, Standard Technical Specifications for Combustion Engineering Plants.

Date of Issuance: February 9, 1995.

Effective Date: February 9, 1995.

Amendment No.: 71.

Facility Operating License No. NPF–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 31, 1994 (59 FR 45024) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: July 25, 1994.

Brief description of amendments: These amendments implement GL 93–05 Items 5.8, 6.1, 7.1 and 7.5.

Date of Issuance: February 9, 1995.

Effective Date: February 9, 1995.

Amendment Nos.: 133 and 72.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 31, 1994 (59 FR 45023) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas.

Date of amendment request: November 7, 1994.

Brief description of amendments: The amendments permit both containment personnel airlock doors to be open while moving fuel during refueling operations.

Date of issuance: February 2, 1995.

Effective date: February 2, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 69; Unit 2—Amendment No. 58.

Facility Operating License Nos. NPF–76 and NPF–80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63123). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: November 8, 1994.

Brief description of amendments: The amendments permit the substitution of an extended range neutron flux monitor for one of the source range neutron flux monitors during refueling operations.

Date of issuance: February 13, 1995.

Effective date: February 13, 1995.

Amendment Nos.: Unit 1—Amendment No. 70; Unit 2—Amendment No. 59.

Facility Operating License Nos. NPF–76 and NPF–80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63124). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: June 6, 1994, as supplemented by letters dated November 17, 1994, and December 5, 1994.

Brief description of amendments: The amendments modify Technical

Specification 3/4.8.1.1, ''A.C. Sources'' by revising the action statements and surveillance requirements for testing of the standby diesel generators (SDGs). The amendments eliminate excessive and unnecessary testing of the SDGs consistent with the guidance provided in NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements,'' NUREG–1431, ''Standard Technical Specifications for Westinghouse Plants,'' Generic Letter 84–15, ''Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,'' and Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' The changes include: (1) eliminating the requirement to demonstrate the operability of an operable SDG whenever an offsite AC power source is determined to be inoperable, or whenever a support system or an independently testable component of another SDG is inoperable, (2) eliminating the requirement to load the diesel in 10 minutes during testing, (3) replacing the minimum required loading for testing with a load band, (4) relocating some surveillance requirements to the Diesel Fuel Oil Testing Program, and (5) eliminating unnecessary loss-of-offsite power tests.

Date of issuance: February 2, 1995.

Effective date: February 2, 1995, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 68; Unit 2—Amendment No. 57.

Facility Operating License Nos. NPF–76 and NPF–80. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37073). The November 17, 1994, and December 5, 1994, submittals provided clarifying information and did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room Location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: August 3, 1994.

Brief description of amendments: The amendments relocate the Radiological Effluent Technical Specifications to other controlled documents consistent with NRC Generic Letter 89–01.

Date of issuance: February 10, 1995.

Effective date: February 10, 1995.

Amendment Nos.: 189 and 175.

Facility Operating License Nos. DPR–58 and DPR–74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 9, 1994 (59 FR 55873).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 10, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: October 24, 1994, as supplemented by letter dated December 16, 1994.

Brief description of amendment: This amendment modifies Technical Specifications Table 4.1–3 surveillance requirements for the new emergency feedwater flow instrumentation. Specifically, the currently installed analog feedwater flow transmitters are to be replaced by new, digital-type flow transmitters. The new digital flow emergency feedwater flow transmitters are continuously self-checking and have a recommended calibration interval of 9 years. The licensee will verify flow whenever the system operates and send one transmitter back to the manufacturer for recalibration every refueling outage.

Date of issuance: February 15, 1995.

Effective date: February 15, 1995.

Amendment No.: 147.

Facility Operating License No. DPR–36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63124). The December 16, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 15, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: May 25, 1994, as supplemented September 1, 1994, and January 13, 1995.

Brief description of amendment: This amendment allows (1) entry through an operable personnel air lock hatch to perform surveillance testing, repair an inoperable hatch, or perform other necessary activities inside containment; (2) update plant Technical Specifications to reflect a previous change to the list of containment boundary valves; (3) add a new exception to allow quarterly surveillance testing of the excess flow check valves; (4) add a new exception to allow periodic preventive maintenance on control room ventilation lasting up to 30 minutes per calendar quarter, without a written report of such inoperability; and (5) make related administrative changes to reflect and clarify items 1 through 4 above.

Date of issuance: February 10, 1995.

Effective date: February 10, 1995.

Amendment No.: 146.

Facility Operating License No. DPR–36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 22, 1994 (59 FR 32231). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1995.

No significant hazards consideration comments received: No

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: November 3, 1993.

Brief description of amendment: The amendment revises License Condition 2.C.(4), ''Turbine System Maintenance Program,'' and deletes Technical Specification (TS) 3/4.3.8, ''Turbine

Overspeed Protection System,'' and its associated Bases. The revision to License Condition 2.C.(4) indicates that the requirements of this license condition have been satisfied. The deletion of TS 3/4.3.8 and its associated Bases provides Niagara Mohawk Power Corporation the flexibility to implement the manufacturer's recommendations for turbine steam valve surveillance test requirements. These test requirements will be contained in the Updated Safety Analysis Report.

Date of issuance: February 14, 1995.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 63.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: December 8, 1993 (58 FR 64611). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: April 25, 1994.

Brief description of amendment: The amendment changes the Technical Specifications concerning four related issues: (1) power-operated relief valve and block valve reliability; (2) low-temperature overpressure protection; (3) boron dilution; and (4) shutdown risk management.

Date of issuance: February 15, 1995.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 185.

Facility Operating License No. DPR–65. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27060). The September 21, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 15, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Thames Valley State Technical College, 574 New London Turnpike, Norwich, Connecticut 06360.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: June 18, 1993, as supplemented by letter dated November 23, 1994.

Description of amendment request: The amendment revises the Appendix A Technical Specifications (TS) relating to the Independent Safety Engineering Group. Specifically, the amendment revises the title of TS 6.2.3 from Independent Safety Engineering Group to Independent Technical Reviews, and replaces the requirements for the five person Independent Safety Engineering Group with requirements relating to a technical review program to perform independent technical reviews.

Date of issuance: February 14, 1995.

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 35.

Facility Operating License No. NPF–86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 18, 1993 (58 FR 43927). The licensee's letter dated November 23, 1994, provided a minor revision to the application but does not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, 47 Front Street, Exeter, NH 03833.

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: October 21, 1994.

Brief description of amendments: These amendments add a test exception to allow reactor coolant temperatures up to 212 degrees F during hydrostatic or inservice leak testing while in OPERATIONAL CONDITION 4 without entering OPERATIONAL CONDITION 3.

Date of issuance: February 13, 1995.

Effective date: To be implemented within 30 days from the date of issuance.

Amendment Nos.: 142 and 112.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 22, 1994 (59 FR 66057). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Philadelphia Electric Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: December 9, 1993, as supplemented by letters dated July 5, September 9, October 19, November 15, and December 2, 1994, January 6 and January 23, 1995. The supplemental letters provided clarifying information that did not change the initial no significant hazards consideration determination.

Brief description of amendment: This amendment raises the authorized maximum power level from 3293 MWt to a new limit of 3458 MWt. The amendment also approves changes to the Technical Specifications to implement uprated power operation.

Date of issuance: February 16, 1995.

Effective date: This license amendment is effective as of its date of issuance and is to be implemented prior to startup in Cycle 4.

Amendment No.: 51.

Facility Operating License No. NPF–85. This amendment revised the Technical Specifications and License.

Date of initial notice in **Federal Register**: February 16, 1994 (59 FR 7695). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Philadelphia Electric Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: October 29, 1993.

Brief description of amendments: These amendments eliminate the main steamline radiation monitoring system high radiation trip function for initiating (1) an automatic reactor scram and automatic closure of the main steamline isolation valves, and (2) automatic closure of the main steamline drain valves, main steam and reactor water sample line valves. The amendments also approve the relocation of portions of the information contained in the Bases section.

Date of issuance: February 16, 1995.

Effective date: February 16, 1995.

Amendment Nos. 89 and 52.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 5, 1994 (59 FR 624). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 16, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: May 13, 1994, as supplemented June 24 and September 27, 1994.

Brief description of amendment: The amendment proposes to amend Appendix A of Operating License DPR–18 to revise Section 6.0 ''Administrative Controls'' of the Ginna Technical Specifications (TSs) and would change the title of Senior Vice President, Production and Engineering, include a provision to allow future title changes without license amendment, and implement those changes in NUREG–1431 ''Standard Technical Specification—Westinghouse Plants,'' dated September 1992, by relocating to licensee controlled documents those specifications controlled by regulations and the existing review and audit requirements. The remainder of this amendment request will be reviewed at a later date.

Date of issuance: February 6, 1995.

Effective date: February 6, 1995.

Amendment No.: 58.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37084). The June 24, 1994, submittal provided information which did not change the initial no significant hazards consideration determination. The licensee's submittal of September 27, 1994, limited, but did not change, the licensee's previously requested TS changes of May 13, 1994.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of application for amendments: December 30, 1993, as supplemented by letters dated June 3, 1994, August 25, 1994, and January 3, 19, and 30, 1995.

Brief description of amendments: These amendments will revise TS Table 3.3–1, ''Reactor Protective Instrumentation,'' to allow the use of the source range neutron flux monitors in place of safety related excore monitors in Modes 3, 4, and 5, with the reactor trip circuit breakers open or the Control Element Assembly (CEA) Drive System not capable of CEA withdrawal, for the purpose of monitoring core reactive changes.

Date of issuance: February 13, 1995.

Effective date: February 13, 1995.

Amendment Nos.: 115 and 104.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 28, 1994 (59 FR 49434). The additional information contained in the January 3, 19, and 30, 1995, letters were clarifying in nature, within the scope of the initial notice and did not affect the NRC staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of application for amendments: July 28, 1994, as supplemented by letters dated January 30 and February 13, 1995.

Brief description of amendments: These amendments propose to revise Technical Specification (TS) 3.9.8.1 ''Shutdown Cooling and Coolant Circulation—High Water Level,'' TS 3.9.8.2 ''Shutdown Cooling and Coolant Circulation—Low Water Level,'' and their Bases to facilitate testing of low-pressure safety injection system components and permit additional flexibility in scheduling maintenance on the shutdown cooling system.

Date of issuance: February 15, 1995.

Effective date: February 15, 1995.

Amendment Nos.: 116 and 105.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications on a one-time basis for each unit.

Date of initial notice in **Federal Register**: October 12, 1994 (59 FR 51627). The additional information contained in the supplemental letters dated January 30 and February 13, 1995, served to clarify the amendments, was within the scope of the initial notice, and did not affect the Commission's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 15, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: October 17, 1994, as supplemented January 30, 1995.

Brief description of amendment: The amendment changes the Technical Specifications to relocate the seismic monitoring instrumentation (SMI) Limiting Condition for Operation (LCO), Surveillance Requirements (SRs), and associated tables and bases contained in Technical Specifications (TS) sections 3.3.3.3 and 4.3.3.3 to the Final Safety Analysis Report (FSAR) or an equivalent controlled document.

Date of issuance: February 15, 1995.

Effective date: February 15, 1995.

Amendment No.: 122.

Facility Operating License No. NPF–12. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: November 8, 1994 (59 FR 55717). The January 30, 1995, supplement did not affect the staff's finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 15, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: September 29, 1993.

Brief Description of amendment: The proposed changes revise standards for testing of charcoal used for removal of radioactive iodine in ventilation systems at the Browns Ferry Nuclear Plant.

Date of issuance: February 13, 1995.

Effective Date: February 13, 1995.

Amendment Nos.: 215, 231 and 188.

Facility Operating License Nos. DPR–33, DPR–52 and DPR–68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 22, 1993 (58 FR 67862). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room Location: Athens Public library, South Street, Athens, Alabama 35611.

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of application for amendment: March 29, 1994.

Brief Description of amendment: The amendment adds requirements for load shedding components being added to ensure that emergency diesel generators are not overloaded during design basis accidents.

Date of issuance: February 14, 1995.

Effective Date: February 14, 1995.

Amendment No.: 189.

Facility Operating License No. DPR–68: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 3, 1994 (59 FR 39597). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room Location: Athens Public library, South Street, Athens, Alabama 35611.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: December 16, 1994; supplemented January 19, 1995 (TS 94–16).

Brief description of amendments: The amendments remove the 900 rpm emergency diesel generator surveillance test criteria and a requirement that the plant be shutdown for performance of the interdependence diesel generator tests.

Date of issuance: February 9, 1995.

Effective date: February 9, 1995.

Amendment Nos.: 195 and 186.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** December 29, 1994 (59 FR 67350). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By March 31, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to *(Project Director):* petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

Commonwealth Edison Company, Docket No. 50–374, LaSalle County Station, Unit 2, LaSalle County, Illinois

Date of application for amendment: January 30, 1995.

Brief description of amendment: The amendment adds a footnote to Technical Specification Table 4.3.1.1–1 to allow a one-time extension of the surveillance interval for the main steam line isolation valve (MSIV) closure reactor protection system channel functional test. This extension averts the need to perform the functional test prior to the start of the upcoming Unit 2 refueling outage.

Date of Issuance: February 15, 1995.

Effective date: Immediately and shall be implemented prior to 2:45 a.m. CST on February 15, 1995.

Amendment No.: 86.

Facility Operating License No. NPF–18: The amendment revised the Technical Specifications.

Press release issued requesting comments as to proposed no siginificant hazards consideration: Yes. February 6, 1995, *Morris Daily Herald; Ottawa Daily Times; and Streator Times-Press.*

Comments received: No. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Illinois and final determination of no significant hazards consideration are contained in a Safety Evaluation dated February 14, 1995.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348.

NRC Project Director: Robert A. Capra.

Dated at Rockville, Maryland, this 21st day of February 1995.

For the Nuclear Regulatory Commission.

John N. Hannon,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–4870 Filed 2–28–95; 8:45 am]

BILLING CODE 7590–01–P

[Docket Nos. 50–245, 50–336, 50–423]

Northeast Utilities; Issuance of Director's Decision Under 10 CFR 2.206

[Millstone Nuclear Power Station] [License Nos. DPR–21, DPR–65, NPF–49]

Notice is hereby given that the Acting Director, Office of Enforcement, has issued a decision concerning the Petitions filed by Ms. Carmela V. Marien and Ms. Marianne W. Nericcio on August 21, 1993. The Petitions requested that the NRC initiate an investigation and accelerated enforcement action against Northeast Utilities (Licensee) for willful violation of the employee protection provisions of 10 CFR 50.7.

After due consideration of Petitioner's assertions, the Acting Director, Office of Enforcement, has denied the Petitions. The reasons for the denial are explained in the ''Director's Decision under 10 CFR 2.206'' (DD–95–04) which is available for public inspection in the Commission's Public Document Room at 2120 L Street NW., Washington, DC 20555.

A copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206. As provided by this regulation, the decision will constitute the final action of the Commission 25 days after the date of issuance of the decision unless the Commission on its own motion institutes a review of the decision within that time.

Dated at Rockville, Maryland this 22nd day of February 1994.

For the Nuclear Regulatory Commission.

Joseph R. Gray,

Acting Director, Office of Enforcement.

[FR Doc. 95–4978 Filed 2–28–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–219]

GPU Nuclear Corporation; Notice of Issuance of Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 177 to Facility Operating License No. NPF–16 issued to GPU Nuclear Corporation (the licensee), which revised the Technical Specifications for operation of the Oyster Creek Nuclear Generating Station located in Ocean County, New Jersey. The amendment is effective as of the date of issuance, to be implemented within 60 days of issuance.

The amendment revises Technical Specification 2.3.D to change the setpoints ''Reactor High Pressure, Relief Valve Initiation'' by increasing the setpoint value by 15 psig for each of the Electromatic Relief Valves in the Automatic Depressurization System.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the **Federal Register** on July 5, 1994 (59 FR 34453). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (60 FR 9056).

For further details with respects to the action see (1) The application for amendment dated June 15, 1994, as supplemented September 23, and November 23, 1994, (2) Amendment No. 177 to License No. DPR–16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Dated at Rockville, Maryland, this 21st day of February 1995.

For the Nuclear Regulatory Commission.

Alexander W. Dromerick, Sr.

Project Manager, Project Directorate I–4, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–4977 Filed 2–28–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–483]

Union Electric Company; Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF–30, issued to Union Electric Company (the licensee), for operation of the Callaway plant, located in Callaway County, Missouri.

The proposed amendment would modify Technical Specification (TS) Section 3/4.9.1 to establish administrative controls to address a possible boron dilution event directly from the reactor makeup water (RMW) system. An unreviewed safety question was involved with the use of RMW to rinse items removed from the refueling pool and to spray down the refueling pool walls during the pool drain evolution. The use of RMW in prior refueling outages during these Mode 6 activities raised the possibility of a different type of accident than any previously evaluated in the Callaway Final Safety Analysis Report (FSAR).

FSAR Section 15.4.6.2 currently states that administrative controls during Mode 6, i.e., closing and locking dilution source manual valves, preclude an inadvertent dilution of the boron concentration of the primary system. Since these valve closures do not preclude the potential dilution scenario described above, different procedural controls are required to ensure that LCO 3.9.1 boron concentration limit of 2000 ppm is met.

NRC Generic Letter 85–05, ''Inadvertent Boron Dilution Events,'' January 1985, and NSAC–183, ''Risk of PWR Inadvertent Criticality During Shutdown and Refueling,'' dated December 1992, documents the technical justification for determining that boron dilution events are self-limiting. Based on the analyses provided in these documents, the staff's acceptance criteria remains valid for the different boron dilution transient (i.e., that gradual boron dilution events are